NON-PUBLIC?: N
ACCESSION #: 8806080417

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Rancho Seco Nuclear Generating Station PAGE: 1 of 9

DOCKET NUMBER: 05000312

TITLE: Automatic Reactor Trip Due To High Reactor Coolant System Pressure
EVENT DATE: 05/04/88 LER #: 88-008-00 REPORT DATE: 06/01/88

OPERATING MODE: N POWER LEVEL: 028

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Dave Schumann, Independent Investigation/Reviews Group
TELEPHONE #: 916-452-3211

COMPONENT FAILURE DESCRIPTION:
CAUSE: D SYSTEM: CB COMPONENT: RV MANUFACTURER: D243
REPORTABLE TO NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: On May 4, 1988, at 09:25:35 hours during the performance of Special Test Procedure STP.660 "Integrated Control System Tuning At Power," the Control Room operators initiated a manual turbine trip from 28% power. Twenty-two seconds later, the reactor tripped automatically on high pressure. At 09:26:49 hours the Main Generator Output Circuit Breakers (OCBs) opened. The delay for the OCB trip was approximately 29 seconds beyond the normal 45-second duration.

At 0928 hours the Control Room operators received initial indications of a leak within the containment. By 0934 hours Control Room operators determined that the "A" Letdown Cooler relief valve (PSV-22021) had lifted and not reseated. By 0940 hours Control Room operators had isolated "A" Letdown Cooler and terminated the leak. Based on an evaluation of the RCS leakage rate, the Shift Supervisor declared and terminated an Unusual Event at 0947 hours. The Shift Supervisor closed out the Unusual Event based on the earlier termination of the leak.

An automatic actuation of the Reactor Protection System resulted in the

unplanned reactor trip and is reportable under the requirements of 10 CFR Part 50.73(a)(2)(iv).

(End of Abstract)

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Description of the Event

o Event Summary

On May 4, 1988, at 09:25:35 hours during the performance of Special Test Procedure STP.660 "Integrated Control System Tuning At Power," the Control Room operators initiated a manual turbine trip from 28% power. Twenty-two seconds later, the reactor tripped automatically on high pressure. At 09:26:49 hours the Main Generator Output Circuit Breakers (OCBs) opened. The delay for the OCB trip was approximately 29 seconds beyond the normal 45-second duration.

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An automatic actuation of the Reactor Protection System resulted in the unplanned reactor trip and is reportable under the requirements of 10 CFR Part 50.73(a)(2)(iv).

o Sequence of Events

The following is a detailed sequence of events:

(All times are approximate except where the Interim Data Acquisition and Display System (IDADS), a computer monitoring system, indicates exact time in seconds)

May 4, 1988

Time Event

0918 Dispatcher notified of intent to trip the turbine

0922 Page announcement of intent to trip the turbine

0925 Second page announcement of intention to trip the turbine

Time Event (Continued)

0925:35 Turbine manually tripped per STP.660, "Integrated Control System Tuning at Power"

0925:48 Atmospheric Dump Valve (ADV) PV-20571A opens

0925:50 ADV PV-20571A closes

0925:51 ADV PV-20562A opens

0925 Control Room operator positions backup pressurizer spray valve in manual and full open (100%); primary spray valve in auto and full open (40%)

0925:56/57 Automatic reactor trip; high RCS pressure (all four RPS channels in trip)

0925:58 ADV PV-20562A closes

0926 "B" HPI pump started manually

0926 Procedure E.01 "Immediate Actions" completed

0926:49 Automatic trip of Main Generator Output Circuit Breakers (OCBs)

0928 Reactor Building Zone 21 fire alarm

0929 Reactor Building Emergency Coolers (C&D) started manually

0930 Reactor Building activity alarm on monitor R-15100

0931:30 First Reactor Building Drain Accumulator Tank (RBDAT) Level Alarm (Hi Hi) (automatic dump on Hi Hi alarm)

0931 Steam observed on Control Room monitor for Reactor Building camera

0932 Reactor Building pressure increases to 1 psig

0933 Procedure E.02 "Vital System Status Verification"
completed; reactor at hot shutdown

0933 Second RBDAT automatic dump

Time Event (Continued)

0934 Control Room operators determine that "A" Letdown Cooler
relief valve may have lifted

0935 Control Room begins preparations to place alternate Letdown
Cooler in service; Control Room operators estimate 30 gpm
leak rate based on RBDAT dumps

0940 "A" Letdown Cooler isolated and alternate Letdown Cooler
placed in service

0947 Shift Supervisor declares Unusual Event based on RCS
leakage rate > Tech Spec Limit and closes out event due to
leak being secured

0954 Notification form read during conference call to
county/state contacts

0956 Leak rate re-evaluated to have been 20 gpm

1002 Control Room starts B&C Reactor Building upper dome air
circulators for temperature considerations

1002 Reset Zone 21 fire alarm; alarm clears

1010 Initiate management personnel notification and closeout on
event per Emergency Plan Implementing Procedure EPIP-5010

1014 Red phone notification to NRC

1015 Initiate post trip review per Administrative Procedure AP.28

1035 Management notification per EPIP-5010 complete

Plant Operating Conditions Before the Event

The plant was at 28% power with generator output at 263 MWe. "A" Letdown

Cooler was in service. The ICS was in automatic except for Delta T(c) and the main feed pumps. "A" main feed pump was in service and in manual control. "B" main feed pump was at minimum speed in manual. STP.660 was in progress under observation in the Control Room by plant management and the NRC resident inspector.

Cause of the Event

The team assembled for the post trip review identified the causes of the following events:

o The RCS High Pressure Trip

o The "A" Letdown Cooler Relief Valve Lift

o The Delayed Trip of the Main Generator OCBs

Each of the following sections provides the conclusions of the team's assessment.

o RCS High Pressure Trip

At 09:25:56/57 the reactor automatically tripped on high RCS pressure. Post trip analysis of the transient data showed that the direct cause of the high pressure condition was a reactor power to feedwater flow mismatch.

When the turbine was tripped, the Integrated Control System (ICS) performed a reactor power and feedwater flow runback. The reactor power runback has a constant speed limited by control rod speed (approximately 20% power per minute), whereas the feedwater flow runback is variable based on the amount of demand from ICS. ICS contains a turbine header pressure compensation circuit which gave the feedwater valves a high closure demand due to the secondary pressure increase when the turbine was tripped. This resulted in the reactor power to feedwater flow mismatch. The initial investigation into the transient established that the ICS responded within its design parameters to the turbine trip. The feedwater flow demand rampdown setting may have been too steep to prevent a reactor trip, but for this setting the ICS and the plant responded correctly to the turbine trip.

Secondary pressure control by the Turbine Bypass Valves (TBVs) also appears to have contributed to the reactor trip. The capacity of the TBVs was within analytical limits, but determined to be less than optimal. Additional TBV capacity would likely have delayed, but not prevented, a reactor trip.

Babcock & Wilcox computer simulations of the turbine trip with Rancho Seco's plant conditions have shown that the reactor trip was a reasonable response to the turbine trip.

o "A" Letdown Cooler Relief Valve Lift

During the transient the relief valve for Letdown Cooler "A" opened and did not reseat. The relief valve was isolated by operator action in the Control Room. Investigation revealed that a discrepancy existed with the setpoint adjustment of the valve at ambient temperature conditions versus normal operating temperature. Additional factors, when added to the temperature offset, establish the potential for a sufficiently reduced relief valve setpoint which could have resulted in a valve actuation during the turbine trip event. A combination of the reduced lift setpoint, plus a 15-20% valve blowdown value identified by the vendor, is the apparent cause of the failure of the valve to reseat.

Prior to the event, in March 1988, "A" Letdown Cooler relief valve PSV-22021 was specified to be factory set at 2500 plus or minus 25 psig for saturated temperature conditions. Prior to installation the valve was bench tested onsite and found to have a lift point of 2580 psig. The valve was then reset to 2480 psig per the Process Standard using procedure MT.006 under ambient shop conditions. Although MT.006 specifies using normal system operating temperature and pressure for in-service testing, the relief valve bench testing section does not specifically provide for temperature compensation or a requirement for a relief media at system operating conditions. According to vendor information, the result of the setpoint adjustment to 2480 psig at cold conditions indicates that the valve may actually lift as low as 2395 psig at RCS temperature. This is due to the setting having been performed at room temperature since the proper cold setpoint has been identified by the vendor as 2575 plus or minus 10 psig. Along with the setpoint change due to temperature, if the elevation difference between the relief valve and system pressure instrumentation is considered, as well as setpoint adjustment and instrument tolerances, there is the potential for a reduced relief valve setpoint. The reduced setpoint could have been sufficient to result in a valve lift and subsequent blowdown during the event.

o Delayed Trip of the Main Generator OCBs

Based on a review of trip data, it was apparent that the OCBs had tripped open 1 minute 14 seconds after turbine trip initiation. The normal delay for OCB trip is approximately 45 seconds after turbine trip initiation. Review of the OCB trip circuit and plant data indicated that the 45 second timer had been reset by procedurally implemented actions after the automatic trip. The

OCB trip circuit functioned as designed.

At 09:25:35 the Control Room operator depressed the turbine trip pushbutton to initiate a turbine trip per STP.660. On release of the pushbutton the 45 second agastat timer began. At 09:25:56/57 the reactor tripped automatically on high pressure. Per immediate actions from procedure E.01 the Control Room operator performed a manual trip of the reactor and turbine. At approximately 09:26:01 the operator depressed the turbine trip pushbutton and on release, the 45 second timer was reset. The OCBs tripped open automatically at 09:26:49, approximately 45 seconds after the second manual turbine trip.

Energy Industry Identification System (EIIS) Component and System Identifies

The system identifiers are JA (Integrated Control System), JC (Plant Protection System), CB (Chemical and Volume Control/Makeup and Purification System) and EL (Main Generator Output Power System). Component identifiers are RV (Relief Valve) and 52 (Circuit Breaker AC).

Manufacturer and Model Number

The manufacturer for PSV-22021, the "A" Letdown Cooler relief valve, is Dresser Industrial Valve and Instrument Division. The model number is 1970. McGraw-Edison Power Systems Division is the manufacturer of the OCBs. The model number is RHE-90-230-20,000.

Method of Discovery

As part of preparations for the turbine trip included in STP.660, arrangements in both personnel briefings and equipment readiness had been accomplished to recover from a potential reactor trip.

For the trip event the Control Room operators noted the high RCS pressure condition. For the relief valve lift, Control Room operators received a fire alarm in Reactor Building Zone 21, visual confirmation via the camera monitor installed in the Control Room, a second RBDAT automatic dump and a Reactor Building pressure increase to 1 psig. For the delayed trip of the Main Generator OCBs the Control Room operators noted that the duration of the delay exceeded the normal period.

Assessment of Safety Consequences

There were no adverse consequences affecting the health and safety of the public or Rancho Seco employees as a result of this event. The discharge from the relief valve lift was ultimately into the Reactor Building sump. Control Room operator action to isolate the "A" Letdown Cooler terminated the discharge. Approximately 1000 gallons of Letdown System inventory discharged to the Reactor Building sump with no offsite release. A component walkdown and sample analysis of the Component Cooling Water System established that there was no damage to the Letdown System or cooler tubes.

Corrective Actions

The following actions were completed prior to reactor restart:

o Update of the Process Standards data base to reflect the correct adjustment conditions and setpoint for the Letdown Cooler relief valves

o Installation of a properly adjusted and tested relief valve on the "A" Letdown Cooler

o Verification of adequately established setpoints on containment relief valves which may require temperature compensation

o Verification of Letdown System integrity via walkdown and sampling of the Component Cooling Water System for activity

o Evaluation of the effect of this transient on the turbine-generator to ensure no adverse equipment consequences

o Adjustment and verification to ensure the proper effect of the turbine header pressure compensation circuit in the ICS

The following actions were completed prior to placement of the Main Generator in service:

o Verification of proper component operation in the OCB trip circuit

o Implementation of a Special Order to ensure operator awareness and appropriate action based on the potential for a timing reset in the Main Generator OCB trip circuit

The following actions will be completed on a post restart basis:

o Evaluation of the automatic positioning of the pressurizer spray valve during reactor trips o Evaluation to optimize TBV capacity

o Vendor examination of the "A" Letdown Cooler relief valve

o Verification of acceptability of the "A" Letdown Cooler relief valve blowdown value

o Performance of a study to determine adequacy of methods for determination of relief valve setpoints

o Evaluation of the trip circuitry for the Main Generator output breakers to assess requirements for changes in operations practices, procedures and subsequent operator training

o Evaluation of the trip circuitry for the Main Generator OCBs to assess requirements for changes in design

o Incorporation of the information concerning Main Generator OCB trip circuitry into operator training and completion of that training by licensed operators

o Completion of STP.660 "Integrated Control System Tuning At Power"

o Review of this event and update of the LER, if required, based on that review

o Evaluation of current methods used to perform technical review of ICS adjustments

Previous Similar Events

LERs 85-19, 85-23 and 85-25 reported reactor trips due to high Reactor Coolant System pressure.

Voluntary LER 88-05 (in review) reported relief valve lifts in the Letdown System due to steam/water hydraulic transients.

ATTACHMENT # 1 TO ANO # 8806080417 PAGE: 1 of 1

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT P. O. Box 15830, Sacramento CA 95852-1830, (916) 452-3211
AN ELECTRIC SYSTEM SERVING THE HEART OF CALIFORNIA

JUN 01 1988

GCA 88-346

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D. C. 20555

Docket No. 50-312
Rancho Seco Nuclear Generating Station
License No. DPR-54
LICENSEE EVENT REPORT 88-08: AUTOMATIC REACTOR TRIP DUE TO HIGH REACTOR
COOLANT SYSTEM PRESSURE

Dear Sir:

In accordance with the requirements of 10 CFR Part 50.73(a)(2)(iv), the Sacramento Municipal Utility District hereby submits Licensee Event Report 88-08.

Members of your staff with questions requiring additional information or clarification may contact Mr. Dave Schumann at (916) 452-3211, extension 4676.

Sincerely,
/s/ G. Carl Andognini
G. Carl Andognini
Chief Executive Officer,
Nuclear

Attachment

cc w/atch:

A. D'Angelo, NRC, Rancho Seco
J. B. Martin, NRC, Walnut Creek
INPO

RANCHO SECO NUCLEAR GENERATING STATION 14440 Twin Cities Road, Herald, CA
95638-9799; (209) 333-2935

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